5. MASS BALANCE ACTIVITIES

5.1 Annual Mass Balance of Recycled Uranium

From startup in FY 1955 through March 31, 1999, approximately 371,000 MTU of uranium (all forms) received at PORTS, with 1,123.7 MTU determined to contain RU. Of this RU, 15.6 MTU are known to have been shipped from PORTS. The balance of the RU was either processed as discussed in other sections of this report or remains in storage.

The annual inventory of RU constituents is shown in Table 5.1-1 and Figure 5.1-1. The data sources are discussed later in Section 5.6

Table 5.1-1

Annual Inventory Of RU Constituents

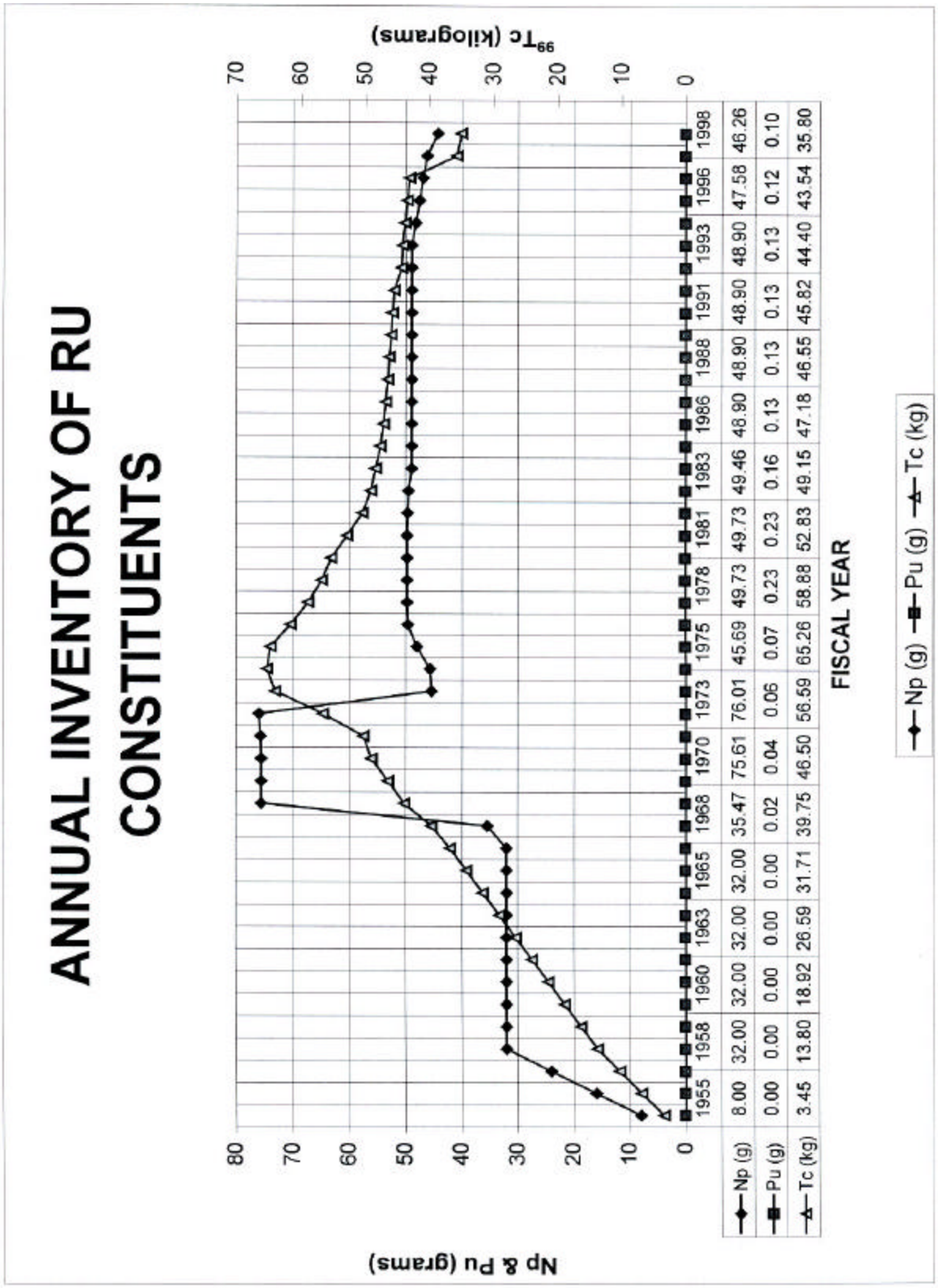
Fiscal Year	Np (g)	Pu (g)	^{99} Tc (kg)	Fiscal Year	Np (g)	Pu (g)	^{99} Tc (kg)
1955	8.00	0.0	3.45	1977	49.62	0.23	61.61
1956	16.00	0.0	6.90	1978	49.73	0.23	58.88
1957	24.00	0.0	10.35	1979	49.73	0.23	56.78
1958	32.00	0.0	13.80	1980	49.73	0.23	55.31
1959	32.00	0.0	16.36	1981	49.73	0.23	52.83
1960	32.00	0.0	18.92	1982	49.66	0.21	50.46
1961	32.00	0.0	21.48	1983	49.46	0.16	49.15
1962	32.00	0.0	24.03	1984	49.90	0.13	48.38
1963	32.00	0.0	26.59	1985	48.90	0.13	47.68
1964	32.00	0.0	29.15	1986	48.90	0.13	47.18
1965	32.00	0.0	31.71	1987	48.90	0.13	46.84
1966	32.01	0.0	34.27	1988	48.90	0.13	46.55
1967	32.02	0.0	36.83	1989	48.90	0.13	46.33
1968	35.47	0.02	39.75	1990	48.90	0.13	46.01
1969	75.61	0.04	43.94	1991	48.90	0.13	45.82
1970	75.61	0.04	46.50	1992	48.90	0.13	45.51
1971	75.61	0.04	49.06	1993	48.90	0.13	44.40
1972	75.72	0.05	50.32	1994	48.90	0.13	44.23
1973	76.01	0.06	56.59	1995	48.24	0.12	43.89
1974	45.42	0.06	63.99	1996	47.58	0.12	43.54
1975	45.69	0.07	65.26	1997	46.92	0.11	43.20
1976	48.03	0.17	64.72	1998	46.26	0.11	35.80
				Mid 1999	44.30	0.11	35.11

5.2 Annual Mass Balance of Plutonium in Recycled Uranium

As discussed in Section 3.1, RU when fed to the cascade loses its identity as RU. However, it is possible to identify the individual processes/facilities, which concentrate isotopes of Np, Pu, and ^{99}Tc and to estimate the mass flow/balance of the Np, Pu and ^{99} Tc.

This section addresses the annual mass balance of the Pu introduced into the PORTS cascade in RU**.**

Figure 5.1-1


ANNUAL INVENTORY OF RU CONSTITUENTS
Np & Pu (grams)
99Tc (kilograms)
FISCAL YEAR
Np (g)
Pu (g)
Tc (kg)
0
10
20
30
40
50
60
70
80
1955 1958 1960 1963 1965 1968 1970 1973 1975 1978 1981 1983 1986 1988 1991 1993 1996 1998
Np (g) 8.00 32.00 32.00 32.00 32.00 35.47 75.61 76.01 45.69 49.73 49.73 49.46 48.90 48.90 48.90 48.90 47.58 46.26
Pu (g) 0.00 0.00 0.00 0.00 0.00 0.02 0.04 0.06 0.07 0.23 0.23 0.16 0.13 0.13 0.13 0.13 0.12 0.10
Tc (kg) 3.45 13.80 18.92 26.59 31.71 39.75 46.50 56.59 65.26 58.88 52.83 49.15 47.18 46.55 45.82 44.40 43.54 35.80

PORTS used four major campaigns to model the flow of Pu from its arrival on plantsite through various facilities and processes to estimate the quantity of Pu present at a given location and to provide a basis for the annual mass balance (see Figures 3.1-2 through 3.1-5).

Plutonium was first introduced in FY 1955 with feed manufactured by Paducah from HRT/SRT. In the early years (FY 1955 – FY 1967) of the PORTS operation, Pu that remained in the feed cylinder heel after feeding was returned with the cylinder to either Paducah or Oak Ridge. The Pu contained in the cylinder heels is assumed not to be in inventory at year-end. Starting in FY 1968, RU was sometimes stored prior to feeding and some remains in storage as of March 31, 1999. In these instances Pu in the cylinders is included in the year-end inventory.

After cascade feeding, 90% of the Pu is assumed to remain in the cylinder heel. The Pu that does enter the cascade deposits on metallic surfaces within the immediate area of the feed point or in the feed lines. During change-out programs of FY 1958 - FY 1960 and FY 1974 - FY 1983 essentially 100% of the Pu introduced up to that point was considered removed. Due to the solubility of Pu in the decontamination solutions, the transfer of Pu from the equipment to the solution is assumed to be nearly 100%. Approximately 99% of the Pu remains with the U_3O_8 produced from uranium recovery. The balance of the Pu traveled with the raffinate to the X-701B. Trace quantities may remain in feed lines or cascade piping near the feed point.

During process equipment change-outs, the equipment typically was decontaminated soon after it was removed. For purposes of the annual mass balance the Pu is counted as either in the cascade, X-701B sludge, oxide storage, or feed cylinders. Material is always assumed to have been completely processed in the same year the processing started.

The amount of Pu in inventory annually is estimated to a peak at 0.23g in FY 1977 through FY 1981, with 0.11g Pu in inventory as of March 31, 1999. The estimate of annual Pu inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.3 Annual Mass Balance of Neptunium in Recycled Uranium

This section discusses the annual mass balance for the Np that was introduced with RU. This is not an annual mass balance of Np in RU (See rationale for Pu mass balance in Section 5.2) The model for estimating Np is similar to that of Pu with the only difference being that the percentage of Np fed to the cascade is assumed to be 33% with 67% remaining in the cylinder heel.

Like Pu, Np was first introduced in FY 1955 with feed manufactured at Paducah from HRT/SRT. Upon receipt, the material was fed and the cylinders returned to either Paducah or Oak Ridge. Beginning in FY 1968, the cylinders were sometimes held for a period of time before feeding; therefore, in these cases, the Np contained in the cylinders is included in the year-end inventory. During periods when a cylinder is fed and returned to Paducah, the Np in the cylinder is not included in the year-end inventory.

The Np that enters the cascade plates out with essentially all of the Np estimated to plate out on cascade components within a range of +6 cells to –4 cells of the feed point (Ref. 27). While Np may spread gradually over extended timeframes, the literature (Ref. 28) suggests sufficient immobility so that it can be assumed that during the cascade change-outs, FY 1958 - FY 1960 and FY 1974 - FY 1983, the equipment where the Np deposited was replaced and essentially 100% of the Np that was fed, up to this time, was removed. Trace quantities undoubtedly remained and perhaps are still present on surfaces not changed out.

The process equipment was decontaminated with essentially all of the Np going into solution through uranium recovery. The oxide produced was stored for oxide conversion at a later date. Approximately 1% of the Np that was processed through uranium recovery ended up in the raffinate at the X-701B with the remaining 99% in the oxide (U_3O_8).

The amount of Np in inventory annually is estimated to peak at 76.01g in FY 1973, with 44.3g Np in inventory as of March 31, 1999.. The estimated annual Np in inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.4 Annual Mass Balance of Technetium in Recycled Uranium

This section discusses the annual mass balance for ^{99}Tc that was introduced to the PORTS site with either RU or ^{99}Tc-contaminated PPF. This is not an annual mass balance of ^{99}Tc in RU (See rationale for Pu mass balance in section 5.2).

The ^{99}Tc mass balance is developed using the same campaigns discussed earlier for Pu and Np to model the constituent movement after arrival at PORTS.

^{99}Tc was first introduced at PORTS in FY 1955 with feed manufactured by Paducah from HRT/SRT oxide and Paducah or Oak Ridge product feed. Upon receipt, the material was fed and the cylinders returned to Paducah or Oak Ridge. Beginning in FY 1968, cylinders were sometimes held for a period of time before feeding. Any ^{99}Tc contained in the cylinders that were stored is included in the year-end inventory. During periods when a cylinder is fed and returned to Paducah, the ^{99}Tc in the cylinder is not included in the year-end inventory.

During cascade feeding, it is estimated that 90% of the ^{99}Tc enters the cascade with 10% remaining in the cylinder. The ^{99}Tc that enters the cascade initially absorbs on the metal surfaces as it moves up the cascade. While ^{99}Tc is highly mobile and moves quickly to the top of the cascade once equilibrium has been established, it was not unequivocally identified until 1974. This 19-year lag is assumed to be at least in part due to the time it took the ^{99}Tc to reach equilibrium (Ref. 19). Once at equilibrium, additional ^{99}Tc in the feed rapidly traveled from the feed point to the top of the cascade. The migration of ^{99}Tc in the cascade was slowed by the equipment change-out in FY 1958 - FY 1960 when much of the equipment contaminated with ^{99}Tc was removed and decontaminated

The process equipment was decontaminated with essentially all of the ^{99}Tc going into solution through uranium recovery. The oxide produced was stored for oxide conversion at a later date. All of the ^{99}Tc processed through uranium recovery is assumed to end up at the X-701B.

The model (Campaigns 2 & 3) includes ^{99}Tc releases to the environment as identified in Table 2.5-1.

The amount of ^{99}Tc in inventory annually is estimated to peak at 65.26 kg in FY 1975. The ^{99}Tc in inventory as of March 31, 1999 is estimated to be 35.11 kg. The estimated annual ^{99}Tc in inventory at PORTS is shown in Figure 5.1-1 and Table 5.1-1.

5.5 Potential for Worker Exposure from Recycled Uranium

Worker monitoring began in 1954 with the Film Badge and Bioassay Programs. Workers with the potential for external radiation exposure were provided film badges for monitoring. However, not all workers were provided film badges, and not all badges issued to workers were read. This changed in the mid 70's when the film badges were replaced with TLD badges. All workers, regardless of exposure potential since that time, have been provided TLD badges. Some badges are not read unless there is cause to believe a significant dose may have been recorded. Records of badge readings obtained since 1954 are retained by USEC.

The bioassay program began with urine sampling for uranium or gross alpha. Uranium sampling was used to monitor intake of workers with the potential for exposure to low assay soluble uranium. Workers with the potential for exposure to high assay uranium were monitored by gross alpha. In the mid-1990s, both methods were replaced with Inductively Coupled Plasma/Mass Spectroscopy (ICP/MS) methods. Results of urine bioassay monitoring since 1954 are retained by USEC.

Since 1988, internal doses have been reported for workers with positive bioassay results (>20 dpm/l) that resulted in calculated doses that exceeded 10 mrem. Up to several hundred employees per year were assigned internal doses based on the alpha bioassay results due to the 20 dpm/l detection limit. When the ICP/MS method replaced the uranium and alpha methods, the workers assigned internal doses dropped to a few each year since the method detection limit is about 10 nanograms (ng)/l for each isotope of uranium. The lower detection limit resulted in fewer workers being assigned doses from false positives. The uranium method (fluorimetry) used previously had a detection limit of about 5 ug/l.

5.5.1 Derived Air Concentration (DAC) and Maximum Permissible Concentrations (MPC)

Over the years of plant operation, the radiation standards have changed. The most recent standards are 10 CFR 20 (USEC) and 10CFR 835 which replaced DOE order 5480.11. Both of these used the DAC based on International Commission on Radiation Protection (ICRP) 26/30 recommendations. The DAC is defined as the concentration that if breathed by a worker for a work-year, would result in a limiting dose. The limiting dose is the more limiting of either 5 rem committed effective dose equivalent or 50 rem committed organ dose equivalent. DACs are listed by each radioisotope and by solubility. The solubility classes, from most to least soluble, are D (for days), W (for weeks), and Y (for years).

AEC/ERDA/DOE 0524 provided Radiation Protection Guidelines (RPGs) based on ICRP 2/10. The MPC if breathed by a worker for a work-year would result in 15 rem annual organ dose. Reporting of internal dose was required when cumulative intakes exceeded 50% of the RPG for the critical organ.

AEC/ERDA/DOE 0524 listed the MPCs for soluble uranium as $6x10^{-11}$ uCi/ml and $1x10^{-10}$ uCi/ml for insoluble uranium. The MPC's for TRU ranged from $1x10^{-10}$ uCi/ml for insoluble ^{237}Np, to $2x10^{-12}$ for soluble ^{238}Pu, ^{239}Pu, and ^{240}Pu. The current DACs from 10CFR 835 are listed in Table 5.5.1-1.

Table 5.5.1-1

Current Derived Air Concentration from 10CFR835

	Half Life (Years)	Specific Activity uCi/g	DACs		
			Class D uCi/ml	Class W uCi/ml	Class Y uCi/ml
^{99}Tc	213,000	17,000	$2x10^{-6}$	$3x10^{-7}$	
^{228}Th	1.91	$8.2x10^{8}$		$4x10^{-12}$	$7x10^{-12}$
^{230}Th	77,000	20,200		$3x10^{-12}$	$7x10^{-12}$
^{234}U	244,500	6,253	$5x10^{-10}$	$3x10^{-10}$	$2x10^{-11}$
^{235}U	7,038,000	2.163	$6x10^{-10}$	$3x10^{-10}$	$2x10^{-11}$
^{236}U	23,415,000	64.74	$6x10^{-10}$	$3x10^{-10}$	$2x10^{-11}$
^{238}U	$4.468x10^{9}$	0.3364		$3x10^{-10}$	$2x10^{-11}$
^{237}Np	2,140,000	705.3		$2x10^{-12}$	
^{238}Pu	87.74	$1.7x10^{7}$		$3x10^{-12}$	$7x10^{-12}$
^{239}Pu	24,065	62,200		$2x10^{-12}$	$6x10^{-12}$
^{241}Am	432.2	$3.4x10^{6}$		$2x10^{-12}$	

The most dramatic change between the old standards and the new standards are for the insoluble TRU DACs. For instance, the MPC for insoluble ^{237}Np was 50 times higher than the current DAC. Since the insoluble TRU MPCs were similar or higher than the insoluble uranium oxide MPC, exposure to insoluble TRU under the old limits would be considered to be adequately controlled if the exposure to insoluble uranium was controlled. The plant allowable limits (PAL) were about half the DOE/Energy Research and Development Administration limits.

To significantly decrease the effective DAC 10% for a mixture of Class D uranium and TRU compared to the uranium alone would require that only 0.04% of the total activity be TRU. The presence of the TRU increases the dose compared to the dose that would have been received by inhalation of the uranium alone. For Class W, there would only have to be 0.07% TRU present for the same 10% decrease in the DAC. For class Y uranium, there would have to be 1.2% present to result in 10% increase in dose.

For ^{99}Tc, the DAC is much higher than the DAC for other beta emitters present at the site. The ^{238}U decay products ^{234}Th and ^{234}Pa are present, especially at the feed facilities. The DAC for ^{234}Th Class Y (the most restrictive) is $6x10^{-8}$ uCi/ml compared to the Class W ^{99}Tc DAC of $3 x10^{-7}$ uCi/ml. Since the uranium alpha DACs are at least 500 times lower, unless ^{99}Tc is present at 500 times the uranium activity, the uranium is the more limiting in terms of the hazard to workers. ^{230}Th is also present at PORTS in significant quantities. ^{230}Th arises from the decay of ^{234}U. About 9 uCi of ^{230}Th per year are produced per curie of ^{234}U. The presence of ^{230}Th is not related to RU since it would be present regardless. It is important to note that ^{230}Th is generally present in larger quantities compared to TRU. The ratio is generally 3 parts ^{230}Th to 1 part TRU. The DAC for ^{230}Th is comparable to TRU. Since 1993, internal dose assessments include a contribution from ^{230}Th and ^{234}U. The significance of TRU to the potential dose to workers in this report will not include the contribution from ^{230}Th.

5.5.2 Bioassay Monitoring Results

From 1965, until the early 1990s, a portable In-Vivo counter was employed to monitor lungs of workers for insoluble uranium. The counter was usually brought to PORTS twice per year. The capability to detect ^{99}Tc, and ^{237}Np were added in 1977. The detection limits for this counter were stated as: 100 ug ^{235}U (240 ug RPG limit); 10 mg total uranium (27 mg RPG limit); 1 uCi ^{99}Tc (9 uCi RPG limit); and 0.2 uCi ^{237}Np (17 nCi RPG limit). The most reliable results were for ^{235}U since the gamma emissions were monitored directly. ^{238}U is determined from the ^{234}Th decay product of ^{238}U, and ^{237}Np is based on the ^{233}Pa emissions. Naturally occurring radon daughters provide a false positive for ^{237}Np. ^{99}Tc is not a gamma emitter, thus a background adjustment is performed over a low energy range to obtain the ^{99}Tc count.

No records have been found that indicate that doses from uranium, ^{99}Tc or TRU have been assigned based solely on In-Vivo results. Usually, another In-Vivo count and a urine bioassay are requested if a result that exceeds the detection limit was obtained. To determine dose, the amount inhaled and how fast the material is removed from each organ of interest must be known. The RPG's are based on having the RPG present in the organ for a year. The results of In-Vivo monitoring were summarized in a 1986 report (Ref. 29.) The following table lists the number of counts taken and the number that exceeded the ^{235}U detection limit from 1965 to 1985: (Table 5.5.2-1)

Table 5.5.2-1

In-Vivo Summary (1965 – 1985)

Year	Total Counts Taken	No. of Counts >100ug^{235}U	Year	Total Counts Taken	No. of Counts >100ug^{235}U
1965	27	18	1976	411	58
1966	30	14	1977	971	96
1967	236	28	1978	542	29
1968	364	39	1979	497	15
1969	393	73	1980	924	4
1970	147	32	1981	868	2
1971	179	56	1982	910	1
1972	157	36	1983	632	3
1973	392	26	1984	613	0
1974	521	65	1985	798	4
1975	684	92			

Another summary produced at the same time (Ref. 30) listed the urine sampling results for uranium and alpha for the same period. The total collected each year and the number that exceeded the detectable level for alpha results (approximately 50 dpm/l) are summarized below: (Table 5.5.2-2)

Table 5.5.2-2

Urine Sampling Summary 1965 - 1985

Year	Total Samples	Positive Samples	% Positive	Year	Total Samples	Positive Samples	% Positive
1965	2,155	1,234	57	1976	5,066	3,689	73
1966	2,667	1,037	39	1977	5,601	4,414	79
1967	2,493	805	32	1978	5,933	2,412	41
1968	2,985	1,035	35	1979	3,694	1,963	53
1969	3,505	1,758	50	1980	7,794	5,525	71
1970	3,442	1,579	46	1981	8,203	5,449	66
1971	3,953	2,121	54	1982	6,571	3,487	53
1972	3,288	1,513	47	1983	4,412	2,455	56
1973	3,656	1,817	50	1984	3,302	1,681	51
1974	3,056	1,489	49	1985	5,108	2,472	48
1975	5,750	3,954	69				

The bioassay results indicate that a large proportion of monitored workers were exposed and had intakes of uranium. The dose to each worker cannot be determined without detailed analysis. Because the monthly samples were not consistently submitted by workers, the actual number of workers monitored each year or the number of workers with positive results can not be reliably extracted from the results above.

5.5.3 Facilities with the Potential for Worker Exposure to RU Constituents

5.5.3.1 X-705 Oxide Conversion Facility

A limited amount of information is available that describes the recycled constituents of the oxide processed in this facility. An unpublished draft report, circa 1977, (Ref. 31) which covered the conversion of TRU contaminated oxides from 1967 to 1975 included calculations of airborne TRU concentration if TRU contaminated oxide were to be processed. This report did indicate that the airborne uranium concentration exceeded the Plant Allowable Level (PAL) in 1368 occurrences in the Tower Room, in 826 occurrences in Oxide Unloading, and an additional 577 occurrences in the Cold Trap Room. The PAL, according to the May 1979 GAT-226 “Guide to Safety”, was 3 dpm/ft^3 ($4.8x10^{-11}$ uCi/ml). The actual data cited in this report could not be located. The use of respiratory protection was required by procedures in place at the time, and their use was encouraged by supervision.

Smear samples obtained in “E”, “F” & “H” areas and analyzed for thorium, uranium and TRU during TRU characterization in the 1990's are shown in Table 5.5.3.1-1.

How much TRU was present in each year of operation is not known however, these sample results do verify that TRU contamination was present when the facility was shutdown in 1978. The samples in H-Area indicate that the TRU percentage is 0.12%. In soluble uranium (Class D or W), found in H-Area and the cold trap room, the levels of RU constituents has a significant effect on the DAC. For TRU in insoluble (Class Y) oxides, (E and F-Areas) the effect is lessened due to the decrease in the insoluble uranium DAC. Since the TRU percentage is less than 1.2%, the effect of TRU in the oxide at these levels is insignificant.

Table 5.5.3.1-1

TRU Characterization of Smear Samples from "E", "F", and "H" Areas

Location	Sample # HPX-	Am/Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
F-Area	93-934	26	32	7,630	0.76	8.18	12.4
H-Area	94-070, 077, 100, 101	3.2	7.5	8,699	0.12	18.86	30.3
E-Area	93-924, 925, 931, 96-039	231	61.0	81,623	0.36	11.66	20.0

Air sampling in the Oxide Conversion Facility measured only the total alpha concentration from uranium, thorium and TRU. Appendix XIV, taken from site internal correspondence (Ref. 32) summarizes the air sample concentration at the continuous air samplers located in the Oxide Conversion Facility from sampler start up through 1978. These levels warrant the use of respirators and, when worn properly, provide adequate protection to the worker.

The calculations for obtaining the DAC are shown in Attachment XIV. The calculated Class D DACs can be compared to the current Class D DAC of $1x10^{-10}$ uCi/ml which assumes that up 2% ^{230}Th is present. The samples from H-Area were used to calculate the DAC for the cold trap room in E-Area, since the TRU fraction is lower and the UF_6 handled is soluble (Class D). This is in line with removal of the TRU in the tower ash and the MgF_2 trap. The airborne radioactivity in the other E-Area samples may contain oxides or ash from the tower which are considered insoluble (Class Y).

5.5.3.2 X-705 Decontamination Area

The X-705 Decontamination Area contains a multitude of activities with the potential for worker exposure to the constituents of RU. There are two principal routes of entry for these constituents - PG cylinders to be cleaned and process equipment to be disassembled and decontaminated. The cylinder cleaning area is in the northwest corner of the X-705. Both large cylinders (2-1/2 ton and up) and small cylinders (5" to 13") are cleaned in separate areas. The cylinders are currently cleaned with a boric acid solution, rinsed and dried. The solutions from the cylinder cleaning are added to the solution recovery system.

Process equipment may be disassembled in one of several areas depending on size and the fixtures necessary to handle it. Gross internal contamination is removed after disassembly. The components are then either placed on carts and passed through the Large Parts Decontamination Tunnel or decontaminated by hand in the Small Parts Area. Solutions from these areas are also added to the solution recovery system.

Air contamination surveys taken in 1993-1994 and 1995-1996 in the X-705 Decontamination Area indicate that significant TRU was present. A summary of these results is shown in Appendix XV.

In the airborne samples taken in 1993-1994, both Pu and ^{237}Np are significant. In the samples taken in 1995 - 1996 only ^{237}Np is significant for the RU constituents. Since the analytical techniques were being refined during 1993 and 1994, it is possible that the difference is due to changes in the laboratory method, or to actual changes in the constituents present.

5.5.3.2-1 Cylinder Cleaning

The cylinder cleaning operations in the X-705 Decontamination Area potentially involve concentrated RU constituents. The results shown in Table 5.5.3.2-1 are from samples taken during decontamination of large cylinders area in 1993. The results in Table 5.5.3.2-2, which were obtained in 1999, were taken from small cylinder wash

solutions in the X-705 West Annex. In both cylinder-cleaning areas, the TRU percentage is significantly greater than 0.04%.

Table 5.5.3.2-1

Large Cylinder Area – Sample Results 1993

Sample # HPX-	Am/Pu pCi	^{237}Np pCi	Total pCi	% TRU	% Th + TRU	uCi/gU	% ^{235}U
93-291, 292, 293, 294, 802, 822, 825, 897, 916	130	60	33,305	0.57	7.94	3.44	4.3

Table 5.5.3.2-2

X-705 West Annex – Sample Results from Small Cylinder Wash

Sample #*	Note	Am/Pu pCi	Np pCi	^{238}Pu pCi	^{239}Pu pCi	Total pCi	% TRU	uCi/gU	% ^{235}U
340-110, 112, 114, 116, 118, 120	Wash	471	11,514	1,287	2,838	1,692,643	0.92	19.82	42.7
340-111, 113, 115, 117, 119	Rinse	0.11	4.8	0.39	1.7	1,305	0.53	17.09	37.1

*Laboratory Information Management System Identification Number (LIMS ID#)

5.5.3.3 X-744G Bulk Storage Building

Batching operations of trap materials were carried out in the X-744G in the mid 1990s. Respirators were worn by workers during these operations. In 1995, air samples were taken and analyzed to characterize the TRU constituents.. The results of all 28 samples taken in 1995 were combined to produce the data in Table 5.5.3.3-1. In this instance the TRU is significant since the TRU percentage is 0.15% of the total activity.

Table 5.5.3.3-1

X-744G Bulk Storage – Air Sample Results for Trap Batching (1995)

^{238}Pu pCi	^{239}Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
1.6	3.9	6.0	7,771	0.15	10.03	16.2

5.5.3.4 X-343 Feed Vaporization and Sampling Facility

This facility feeds UF_6 into the cascade. Airborne radioactivity measurements analyzed for TRU are summarized in Table 5.5.3.4-1. The results were obtained from 38 samples taken between 1994 and 1997. The TRU levels are insignificant at 0.01% of the total activity.

Table 5.5.3.4-1

X-343 Feed Vaporization and Sampling Facility – Airborne Radioactivity Summary (1994 – 1997)

^{239}Pu pCi	^{238}Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
0.0	0	1.2	19,986	0.01	2.10	4.0

5.5.3.5 X-344 Toll Enrichment Facility

The sample results shown in Table 5.5.3.5-1 were obtained from eleven air samples taken in 1994 - 1997 in the X-344. The TRU levels are insignificant at less than 0.01% of the total activity.

Table 5.5.3.5-1

X-344 Toll Enrichment Facility – Airborne Radioactivity Summary (1994 – 1997)

^{239}Pu pCi	^{238}Pu pCi	^{237}Np pCi	Total pCi	%TRU	uCi/gU	% ^{235}U
0.0	0.21	0.7	25,143	0.00	0.95	1.3

5.6 Environmental

Surveys to determine the extent of contamination in the USEC leased facilities covered more than 540 outside acres. Most outside areas covered with grass, gravel or pavement were surveyed with an array of radiation detectors towed behind a tractor at slow speeds. Other areas were surveyed with hand-held survey instruments.

USEC Health Physics Policy X38300-00-001 lists all areas of contamination within the USEC leased areas. One area is known to contain ^{99}Tc and TRU (X-701B), but most are posted only for protection of personnel from removable contamination. There are about 527,000 sq. ft. of contaminated areas, of which almost half (244,000 sq. ft.) are on the roofs of the X-705 and X-710. The X-701B is about 260,000 sq. ft. Contamination control zones are 4,116,800 sq. ft. of which 4,109,000 sq. ft. are associated with the three process buildings. There are 1,532,900 sq. ft. of fixed contamination areas with 530,000 sq. ft. in the X-530 switchyard. Soil contamination areas amount to 199,100 sq. ft. of which 130,000 sq. ft. are near the X-7721, 12,000 sq. ft. near the X-745F, and 14,000 sq. ft. near the X-705. Underground radioactive material areas amount to 15,800 sq. ft. and mostly associated with the X-705 (15,500 sq. ft.).